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Modelling of HTR (High Temperature Reactor) Pebble-Bed 10 MW to Determine Criticality as A Variations of Enrichment and Radius of the Fuel (Kernel) With the Monte Carlo Code MCNP4C

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Abstract - *Gas-cooled nuclear reactor is a Generation IV reactor which has been receiving significant attention due to many desired characteristics such as inherent safety, modularity, relatively low cost, short construction period, and easy financing. High temperature reactor (HTR) pebble-bed as one of type of gas-cooled reactor concept is getting attention. In HTR pebble-bed design, radius and enrichment of the fuel kernel are the key parameter that can be chosen freely to determine the desired value of criticality. This paper models HTR pebble-bed 10 MW and determines an effective of enrichment and radius of the fuel (Kernel) to get criticality value of reactor. The TRISO particle coated fuel particle which was modelled explicitly and distributed in the fuelled region of the fuel pebbles using a Simple-Cubic (SC) lattice. The pebble-bed balls and moderator balls distributed in the core zone using a Body-Centred Cubic lattice with assumption of a fresh fuel by the fuel enrichment was 7-17% at 1% range and the size of the fuel radius was 175-300 µm at 25 µm ranges. The geometrical model of the full reactor is obtained by using lattice and universe facilities provided by MCNP4C. The details of model are discussed with necessary simplifications. Criticality calculations were conducted by Monte Carlo transport code MCNP4C and continuous energy nuclear data library ENDF/B-VI. From calculation results can be concluded that an effective of enrichment and radius of fuel (Kernel) to achieve a critical condition was the enrichment of 15-17% at a radius of 200 µm, the enrichment of 13-17% at a radius of 225 µm, the enrichments of 12-15% at radius of 250 µm, the enrichments of 11-14% at a radius of 275 µm and the enrichment of 10-13% at a radius of 300 µm, so that the effective of enrichments and radii of fuel (Kernel) can be considered in the HTR 10 MW.*

Keywords—*MCNP4C, HTR, enrichment, radius, criticality*

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I. Introduction

Like the fast reactor, the high temperature gas-cooled reactor using helium as the coolant and graphite as the moderator, has exerted a peculiar attraction over scientist. There are a number of current design studies for different type of gas-cooled power reactors. One type of gas-cooled power reactors is High Temperature Reactor (HTR) which is getting attention. One type of HTR is HTR fuel pebble-bed. The HTR utilizes graphite as the moderator at the same reflector and the fuel is a spherical particle (pebble-bed) with UO_2 composition as a neutron generator. HTR-10 is a high temperature reactor power of 10 MW which is still operated and developed in China. Various studies about HTR have been carried. HTR-10 core design is a cylindrical with helium gas as a coolant and graphite as a moderator. In addition, the HTR-10 uses pebble-bed fuel composed a large amount of particles of TRISO in a graphite metrics. The TRISO (tri structural isotropic) particle is coated fuel particle by a radius of 175-300 µm. HTR pebble-bed reactor required fuel enrichments of 7-10% for reach criticality condition. Suwoto's research found that the value of the criticality of the reactor will decrease when the enrichment is smaller (Holbrook, 2008; Zuhair, 2012; Suwoto, 2010).

In HTR pebble-bed design, radius and enrichment of the fuel UO_2 is a key parameter that parameter that can be chosen freely to determine the desired value of criticality. The value of criticality reactor is a variable that describes the state of the reactor in order to operate optimally. Therefore, it is needed to simulate the state of the reactor at different radius and enrichment of the Kernel before it is operated in order to produce the desired value of criticality.

MCNP4C program is one of the software that can be used to simulate the process in the reactor core. According to Kim, that the program MCNP simulation can be used as a pebble-bed core, the Kim's research simulate HTR-10 reactor with randomly packing pebble in a radius of 250 μm and a fuel enrichment of 17% with a terrace height of zone fuel at 125 cm was obtained criticality values by 1.00002. this software has some advantages that can be used to model complex geometry and can determine the criticality of a reactor (H.C. Kim, S. H. Kim and J. K. Kim, 2011).

II. MATERIAL AND METHOD

Materials used in this research were HTR-10 database design and continuous energy nuclear data library ENDF/B-VI. HTR-10 is a high temperature reactor power of 10 MW which is still operated and developed in China. The reactor is a pebble-bed HTR utilizing spherical fuel elements with ceramic coated fuel particles and utilizing a gas-cooled. The key advantages of the HTR pebble-bed is located on the fuel loading which can be done online without having to stop the production of electricity. As a consequence, the reactor has a low reactivity. To maintain the reactor in a critical condition, new pebbles of the fuel can be added and removed the old pebble. The reactor core has a diameter of 1.8 m, a mean height of 1.97 m and the volume of 5.0 m and is surrounded by graphite reflector. The core is composed of 27,000 fuel elements.

In the general reactor design, graphite serves as the primary core structural material, which is primarily in the top, bottom and side reflector. The ceramic core structure are housed in a metallic core vessel that is supported on the steel pressure vessel. The thickness of the side reflector is 100 cm; including a layer of carbon bricks. cold helium channels are designed within the side reflector for the primary coolant (helium) to flow upward after entering the reactor pressure vessel from the annular space between the connecting vessel and the hot gas duct. Helium flow reverses at the top of reactor core to enter the pebble-bed, so that a downward flow pattern takes place. After being heated in the pebble-bed, helium then enters into a hot gas chamber in the bottom reflector, and from there it flows through the hot gas duct and then on to the heat exchanging components. Spherical fuel elements (6 cm in diameter) with TRISO coated fuel particles make up the core. These elements go through the reactor core in a multi-pass pattern with a pulse pneumatic fuel handing system utilized for continually charging and discharging the fuel elements (IAEA, 2003).

Cross-section views of the HTR-10 reactor are shown in Figure 1 and 2. In the side reflector near the active core, there are ten borings (diameter = 130 mm) for control rods, seven borings for small absorber balls and three borings (diameter = 130 mm) for irradiation purposes. There are twenty flow channels (borings with a diameter of 80 mm) in the side reflector for reactor inlet helium. The active reactor core containing spherical balls is surrounded by graphite reflectors and the graphite reflectors are surrounded by a layer of borated carbon bricks (IAEA, 2003). Table 1 illustrates the key design parameters of HTR-10.

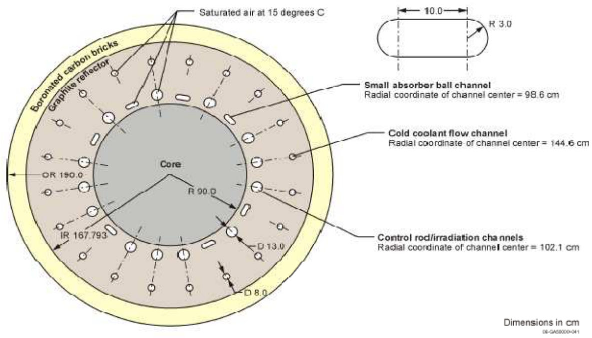

Figure 1. HTR-10 reactor horizontal cross-section (W. Terry, 2006)

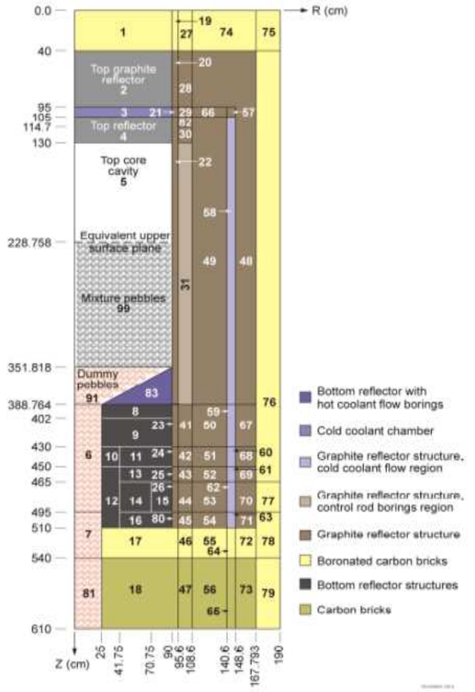

Figure 2. HTR-10 reactor vertical cross-section (W. Terry, 2006)

For the initial core loading, dummy balls (graphite balls without nuclear fuel) will be firstly placed into the discharge tube and the bottom cone region of the reactor core. Then, a mixture of the pebble balls and dummy balls will be loaded gradually to approach first criticality. The percentages of the pebble balls and dummy (moderator) balls are envisaged to be 57% and 43% respectively (IAEA, 2003). Table 2 and Figure 3 illustrate the basic characteristics and the schematic of the fuel elements

Table 1.Key design parameters of the HTR-10 (IAEA, 2003)

Core	
Reactor thermal power (MW)	10
Average core height (m)	1.97
Reactor core diameter (m)	1.80
Pebble packing fraction	0.61
Materials	
Nuclear fuel	UO_2
Coolant	He
Reflector	Graphite
Coolant	
Temperature at reactor inlet/outlet (°C)	250/700
Primary helium pressure (MPa)	250
Helium mass flow rate at full power (kg/s)	4.30

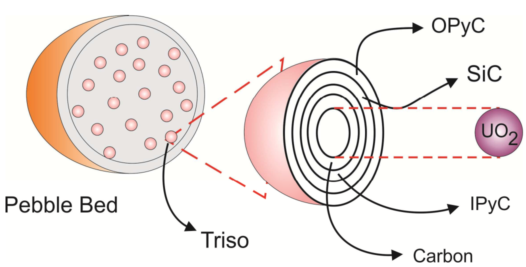

Figure 3. HTR-10 spherical fuel element

Table 2.Fuel element characteristics (IAEA, 2003)

Fuel kernel	
Diameter of ball (cm)	6.0
Diameter of fuelled region (cm)	5.0
Density of graphite in matrix and outer shell (g/cm³)	1.73
Heavy metal (uranium) loading (weight) per ball (g)	5.0
Enrichment of 235U (w%)	7-17
Equivalent natural boron content of impurities in uranium (ppm)	4.0
Equivalent natural boron content of impurities in graphite (ppm)	1.3
Radius of the kernel (µm)	175-300
UO_2 density (g/cm³)	10.4
Coatings	
Coating layer materials (starting from kernel)	C/IPyC/SiC/OPyC
Coating layer thickness (mm)	0.09/0.04/0.035/0.04
Coating layer density (g/cm³)	1.1/1.9/3.18/1.9
Dummy (no fuel) elements	
Diameter of ball (cm)	6.0
Density of graphite (g/cm³)	1.73
Equivalent natural boron content of impurities in graphite (ppm)	1.3

Modelling of HTR pebble-bed uses the Monte Carlo code MCNP4C. MCNP (Monte Carlo N-Particle) is a general-purpose, continuous-energy, generalized-geometry, time-dependent, coupled neutron, photon and electron Monte Carlo transport code. MCNP is also capable of calculating the multiplication factor (criticality) of fissile systems. A system is defined by generating cells bounded by surfaces in three dimensions (J. F. Briesmeister, 1992).

MCNP is also capable of calculating the multiplication factor (criticality) of fissile systems. A system is defined by generating cells bounded by surfaces in three dimensions. Any kind of geometry can be defined as a cell and this cell can be rotated and moved to anywhere in the space. MCNP can simulate particle transport. Monte Carlo simulates individual particles and recording some aspects (tallies) of their average behaviour. Simulations can be performed by either discrete (multi-group) or continuous energy cross-section. ENDF/B-VI continuous energy cross-sections are used in calculations for all materials except graphite. Cross-sections for graphite are taken from TMCCS library.

Criticality evaluations are done based on the principle of neutron balance. The number of neutrons in each generation is taken into account and comparison is made with the number of neutrons in the consequent generation. All possible mechanisms for the birth and loss of neutrons are accounted in bookkeeping. Thus, effective multiplication factor is evaluated for a given cycle. Each fission neutron is generated randomly out of possible locations containing fissile material. In order to generate statistical basis, simulations are repeated as many times as desired (Volkan, 2002).

The initial step is to model the TRISO, the individual TRISO coated fuel particle (UO_2) and four outer layers (Carbon, IPyC (Inner Pyrolytic Coating), SiC (Silicon Carbides) and OPyC (Outer Pyrolytic Coating) which are shown in Figure 4. After modelling the TRISO particle, the next step is to model pebble-bed which has a radius of 3 cm and a radius of the fuelled region of 2.5 cm which is filled by graphite. Then, the TRISO particles were distributed in the fuelled region of the fuel pebbles using a simple-cubic (SC) lattice by packing fraction of 0.05 which is made constant to maintain a constant ratio of graphite and uranium in each radius of kernel which is considered. Figure 5 shows MCNP model for a pebble-bed.

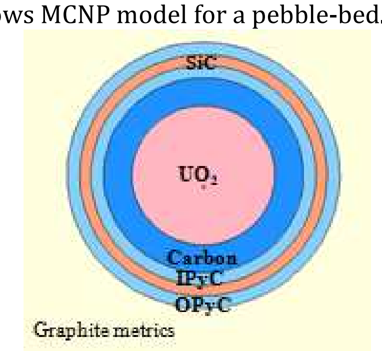

Figure 4. MCNP model of coated fuel particle (TRISO)

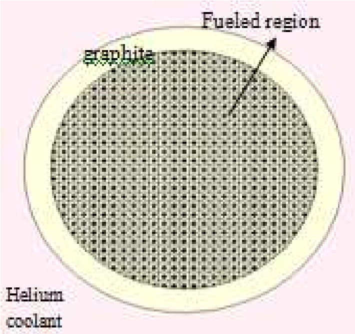

Figure 5. MCNP model of pebble-bed

Pebble-bed balls and moderator balls were distributed in the core zone of the HTR using a body-centred cubic (BCC) lattice by packing fraction and the percentages of the pebble balls and moderator balls of 0.61 and 57:43. Then, the pebble-bed core was modelled by lattice structure utilizing LATTICE option to distributed 24,487 pebbles in body-centred cubic (BCC) lattice. Then, the completely reactor structure is created. The MCNP model consists of the reactor structure which includes the graphite reflector and the borated carbon bricks that surround the reflector and the pebble-bed core. Figure 6 and 7 show cross-section view of the modelled structure.

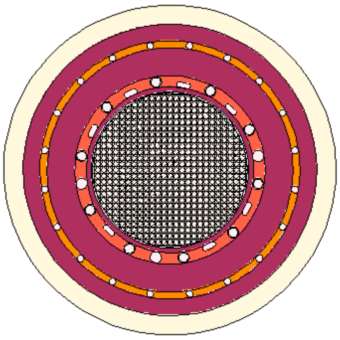

Figure 6. Horizontal cross-section view of the MCNP model of HTR-10

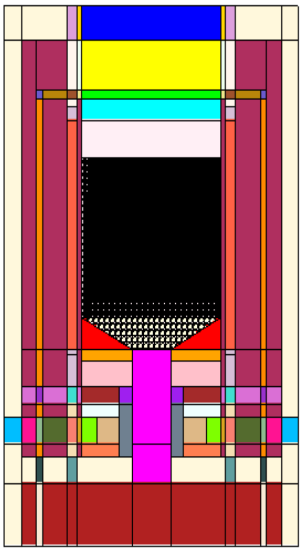

Figure 7. Vertical cross-section view of the MCNP model of HTR-10

III. RESULT AND DISCUSSION

In MCNP calculation, the number of neutron simulating in the KCODE card and neutron source in the SDEF card must be specified. There are 5,000 neutrons in each cycles and initial guesses criticality of 1.0 in the KCODE card that are used in this calculation. Skipping 10 cycles is done before criticality data accumulation from a total of 250 cycles to prevent convergence of the source. Initial estimation criticality value of 1.0 is selected, so that the final accumulation results is expected nearly equal to the criticality condition whereas skipping 10 cycles is used, so that fission sources can be stable before criticality values are used to average its final estimation. SDEF card is utilized to calculation criticality value by describe fission source distributions in all fuels at reactor core. Calculation criticality value is done by continuous energy nuclear data library ENDF/B-VI at temperature of 293K and at high critical core of 178.41 cm. S(α,β) graph.01t of thermal neutron scattering data is applied in all materials containing graphite to consider binding effect at thermal neutron and graphite moderator under energy of 4 eV.

Criticality value affects the state of the reactor operation. If the criticality value is less than one, so the reactor cannot operate because the numbers of neutrons producing are less. If the criticality value is greater than one, so the number of neutrons will continue to increase from generation to generation and if the criticality value is nearly equal to one, as a result of the reactor will be stable condition because the numbers of neutrons are constant from generation to generation. Criticality value will affect the reactivity of the reactor. It describes major changes in the population of neutrons generation to generation and is used to determine the level of criticality.

If the reactivity value is less than zero (negative value) as a result of the reactor will be subcritical condition because the number of neutrons population in next generation are less. If the reactivity value is nearly equal to zero as a result of the reactor will be critical condition because the number of neutrons populations in the next generation are constant. If the reactivity value is greater than zero then the reactor will be supercritical condition because the next generation of neutron populations will increase (DOE, 1993). Criticality and reactivity value in each fuel enrichments and radii are shown in Figures 8 and 9.

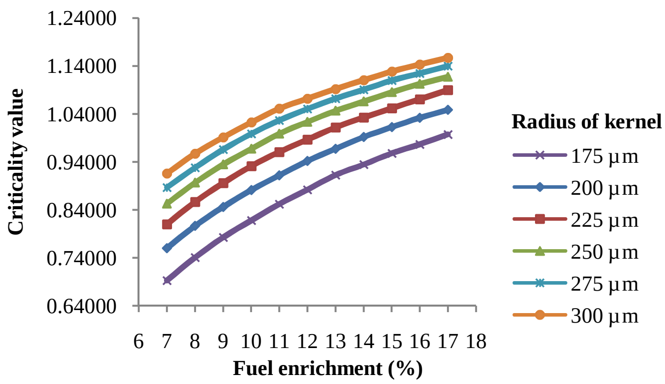

Figure 8. Graph criticality value of reactor with fully out control rod

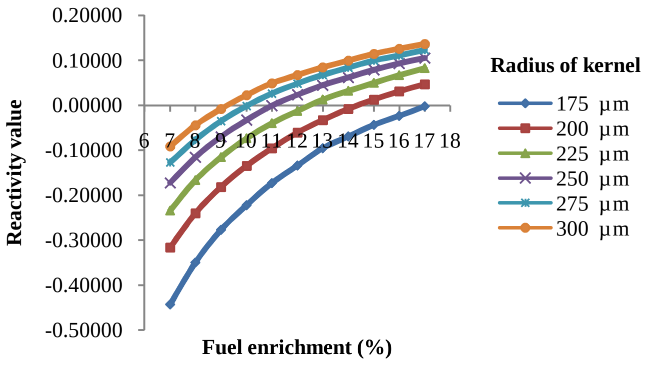

Figure 9: Graph reactivity value of reactor with fully out control rod

From the graph in Figures 8 and 9 that high fuel enrichments cause values of criticality and reactivity increase because there were a large amount of uranium-235 in the reactor core, so that using of thermal neutron to fission will increase. Radius of kernel is small, as a result of the volume of kernel will be smaller, so that the number of atom fuels resulting fission will be less, consequently the criticality value will be smaller. In the reactor design is required an effective of enrichment and radius of the fuel so that the reactor can operate optimally.

From the graph in Figure 8 that the criticality value has not achieved critical condition yet in all fuel enrichments at a radius of 175 µm. In the fuel enrichments of 16-17% at a radius of 250 µm, in the fuel enrichments of 15-17% at a radius of 275 µm and in the fuel enrichments of 14-17% at a radius of 300 µm can be considered to be the fuel in reactor but it is not effective because the value of criticality is greater than one as a result of the number of neutrons will continue to increase from generation to generation and can be called in a supercritical condition.

The effective of enrichments and radii of the fuel to achieve the criticality value is nearly equal to one and reactivity value is nearly equal to zero so that is called in a critical condition, i.e. the enrichments of 15-17% at a radius of 200 µm, the enrichments of 13-17% at a radius of 225 µm, the enrichments of 12-15% at a radius of 250 µm, the enrichments of 11-14% at a radius of 275 µm and the enrichments of 10-13% at a radius of 300 µm.

IV. CONLUSIONS

From this result of research can be got a conclusion that the criticality value has not achieved critical condition yet in all fuel enrichments at a radius of 175 µm and the effective of enrichment and radius of the fuel has already achieved critical condition and can be considered to be the fuel in the HTR 10 MW, i.e. the enrichments of 15-17% at a radius of 200 µm, the enrichments of 13-17% at a radius of 225 µm, the enrichments of 12-15% at a radius of 250 µm, the enrichments of 11-14% at a radius of 275 µm and the enrichments of 10-13% at a radius of 300 µm, so that these enrichments and radii can be considered in the HTR 10 MW.

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